EPJ Web of Conferences **146**, 09005 (2017) DOI: 10.1051/epjconf/201714609005
ND2016

On prediction accuracy of neutronics parameters of accelerator-driven system

Go Chiba[1,a], Tomohiro Endo[2], Wilfred van Rooijen[3], and Cheol Ho Pyeon[4]

[1] Division of Energy and Environmental Systems, Hokkaido University, Sapporo, Japan
[2] Graduate School of Engineering, Nagoya University, Nagoya, Japan
[3] Research Institute of Nuclear Engineering, University of Fukui, Tsuruga, Japan
[4] Research Reactor Institute, Kyoto University, Kumatori, Japan

Abstract. Nuclear data-induced uncertainty of neutronics parameters of one specific ADS design is quantified. The nuclear data adjustment method with available integral data is employed to reduce the uncertainties, and usefulness of these integral data is investigated. Numerical results reveal that the uncertainty reduction by the present nuclear data adjustment is insignificant and restrictive. Future perspecitives are also provided.

1. Introduction

An accelerator-driven system (ADS) is one of promising nuclear systems, which has a potential to drastically reduce the burden of nuclear waste disposal by burning minor actinoid nuclides. Since accurate prediction of neutronics parameters of ADS is essential and important, so much effort have been devoted to quantify and reduce the uncertainties of the ADS neutronics parameters. In the present work, we quantify nuclear data(ND)-induced uncertainty of neutronics parameters of one specific ADS design. This is accomplished with combined information on microscopic data (nuclear data) and macroscopic data (integral data). This combination is realized by using the ND adjustment method.

2. Integral data

At present, several integral data which are related to the ADS neutronics parameters are available. Table 1 shows a list about integral data which are used during the present work. A specific ID is given to each integral data throughout the present paper. From the ICSBEP handbook, several critical data and sample reactivity worth data, which are expected to be sensitive to ND of minor actinoid nuclides and lead isotopes, are chosen. In addition to these ICSBEP-originated data, the reaction rate ratio measurement data obtained at Los Alamos National Laboratory (LANL) are also used. Furthermore, Fukushima et al. has recently prepared the integral benchmark data of fission rate ratio measurements of transuranic nuclides conducted at six cores of Fast Critical Assembly (FCA) in Japan [1,2]. Fission reaction rate ratios of Np-237, Pu-238, -242, Am-241, -243 and Cm-244 against Pu-239 are included in this benchmark problem. Three-dimensional heterogeneous reactor cores are simplified to cell-homogenized ones in the benchmark models.

3. Input data for nuclear data adjustment

In the ND adjustment calculations, several quantities on the integral data are required: calculation values obtained with the original ND and their uncertainties, measurement values and their uncertainties, and sensitivity profiles of calculation values with respect to ND. Covariance data of ND are also required.

3.1. Calculation values with their uncertainties

The present ND adjustment is conducted with JENDL-4.0. Since uncertainties of calculation values should be as small as possible, we use calculation values obtained with continuous-energy Monte Carlo codes if these are available.

Calculation values of k_{eff} except for three data of HMF-064 are taken from the results obtained by Okumura and Nagaya with MVP-II [4], and those of the HMF-064 data are taken from the results obtained by van der Marck with MCNP [5]. Statistical uncertainties of these results are negligibly small, so uncertainties of 0.001% are assumed to all these calculation values.

On the FCA-IX benchmark problems, Fukushima et al. also provide numerical results by MVP-II. Their statistical uncertainties are about 0.1% to 0.2% [3], so uncertainties of 0.2% are assumed.

The sample reactivity worth data and the reaction rate ratio data obtained at LANL are calculated by a deterministic reactor physic code system CBZ which is under development at Hokkaido University. Neutron flux in 175 energy groups are calculated by a discrete-ordinate (S_N) neutron transport solver SNR. 175-group

[a] e-mail: go_chiba@eng.hokudai.ac.jp

Table 1. Integral data with their ID.

ID	ICSBEP ID or core name	Parameter
1	Godiva	k_{eff} (HEU)
2	SMF-008	k_{eff} (HEU+Np)
3	SMF-011	k_{eff} (HEU+Np+Polyethelene)
4	SMF-014	k_{eff} (HEU+Np+Iron)
5	SMF-002	Pu-239 sample worth (SW)
6	SMF-002	Pu-238 SW
7	SMF-001	Cm-244 SW
8	SMF-001	Pu SW
9	SMF-001	U SW
10	SMF-003	Np SW in HEU fuel
11	SMF-003	SW (HEU → Np) in HEU fuel
12	SMF-003	HEU SW in HEU fuel
13	SMF-003	SW (HEU → Np) in Pu fuel
14	Godiva	U-238f/U-235f
15	Godiva	Np-237f/U-235f
16	Godiva	U-233f/U-235f
17	Godiva	Pu-239f/U-235f
18	Jezebel	U-238f/U-235f
19	Jezebel	Np-237f/U-235f
20	Jezebel	U-233f/U-235f
21	Jezebel	Pu-239f/U-235f
22	Jezebel-233	U-238f/U-235f
23	Jezebel-233	Np-237f/U-235f
24	Flattop-25	U-238f/U-235f
25	Flattop-25	Np-237f/U-235f
26	Flattop-25	U-233f/U-235f
27	Flattop-25	Pu-239f/U-235f
28	Flattop-Pu	U-238f/U-235f
29	Flattop-Pu	Np-237f/U-235f
30	Flattop-233	U-238f/U-235f
31	Flattop-233	Np-237f/U-235f
32	HMF-018	k_{eff} (HEU)
33	HMF-027	k_{eff} (HEU+Pb)
34	PMF-022	k_{eff} (Pu)
35	PMF-035	k_{eff} (Pu+Pb)
36	HMF-064-1	k_{eff} (HEU+Pb)
37	HMF-064-2	k_{eff} (HEU+Pb)
38	HMF-064-3	k_{eff} (HEU+Pb)
39-45	FCA-IX-1 to -7	Np-237f/Pu-239f
46-51	FCA-IX-1 to -7	Pu-238f/Pu-239f
52-58	FCA-IX-1 to -7	Pu-242f/Pu-239f
59-65	FCA-IX-1 to -7	Am-241f/Pu-239f
66-72	FCA-IX-1 to -7	Am-243f/Pu-239f
73-79	FCA-IX-1 to -7	Cm-244f/Pu-239f

cross sections are calculated with the 175-group CBZLIB based on JENDL-4.0. Scattering anisotropy is taken into account by the fifth-order Legendre polynomials. Adjoint neutron flux is also calculated, and the sample reactivity worth is evaluated with the exact perturbation theory. Spatial, angle and energy discretizations are so fine that uncertainties of these calculation values are assumed 1.0%.

3.2. Experimental values with their uncertainties

On the integral data taken from the ICSBEP handbook, all the information on experimental data are provided in the handbook. Note that uncertainties of β_{eff} in the sample reactivity worth data are treated as one of experimental uncertainties and are taken from the handbook.

On the reaction rate ratio data at LANL, experimental values and their relative standard deviations are taken from the reference [6]. Any correlations among experimental values are not assumed here.

On the FCA-IX benchmark, experimental values and their relative standard deviations are taken from the reference [1]. Although correlations are not evaluated in this reference, it is mentioned that the main source of experimental uncertainties is the amount of fissile nuclides used in the fission chambers, so relatively strong correlation of 0.8 is assumed among the same fission reaction rate ratios.

3.3. Sensitivity profiles and covariance data of nuclear data

The sensitivity profiles are calculated in 70-group structure with CBZ for all the integral data. Forward and (generalized) adjoint neutron fluxes required for the (generalized) perturbation theory-based sensitivity calculations are obtained as follows. For spherical systems, the SNR solver of CBZ is employed. The three assemblies of HMF-064 have three-dimensional structure, but those are simplified to two-dimensional cylindrical systems in the sensitivity calculations. Neutron fluxes are calculated with a S_N solver SNRZ of CBZ for these systems. On the FCA-IX benchmark, a three-dimensional S_N solver SNT of CBZ is employed.

Covariance data given in JENDL-4.0 are used for the following nuclides: U-233, -234, -235, -238, Np-237, Pu-238, -239, -240, -241, -242, Am-241, -243, Cm-242, -244, -245, -246, N-15, Fe-56, Pb-206, Pb-207, Pb-208, Pb-209, Bi-209. These are processed into the 70-group structure by NJOY-99. Note that non-cross section ND such as $\bar{\nu}$, $\bar{\mu}$ and χ are also considered.

4. Dedicated ADS

We will quantify the uncertainty reduction by the ND adjustment for a commercial grade ADS proposed by the Japan Atomic Energy Agency [7]. This system uses a lead-bismuth eutectic coolant. For the core fuel, mixture of mono-nitride of MA and Pu with the inert matrix zirconium-nitride is used. A proton beam with 1.5 GeV provided by the LINAC is injected into the core through the beam duct along the core central axis. The core thermal power is 800 MW and burnup period is 600 effective full power days. After each burnup cycle, all fuels are removed from the core and reloaded for next burnup cycle after cooling and reprocessing. The time period for the cooling and the reprocessing is 2.5 years in total. In the refabrication process, fission products are removed and only MA of equal mass to the burnup fuel is added to the recycled fuel. Plutonium is loaded as mixture of MA at the first cycle to suppress the burnup reactivity swing.

In the present study, we focus on k_{eff}, β_{eff} and coolant void reactivity. Sensitivities of these parameters with respect to ND are taken from our previous work [8].

5. Numerical results

Figure 1 shows (C/E-1) values of the integral data, which are used for the ND adjustment, normalized by

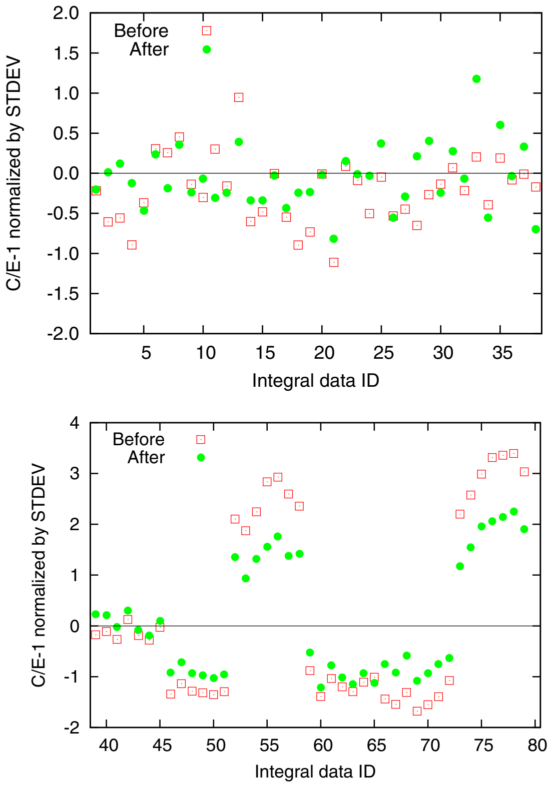

Figure 1. (C/E-1) values of the integral data used for the adjustment normalized by thier relative standard deviations.

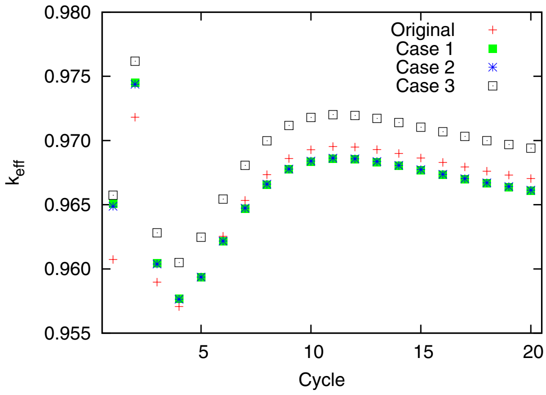

Figure 2. Effective neutron multiplication factors.

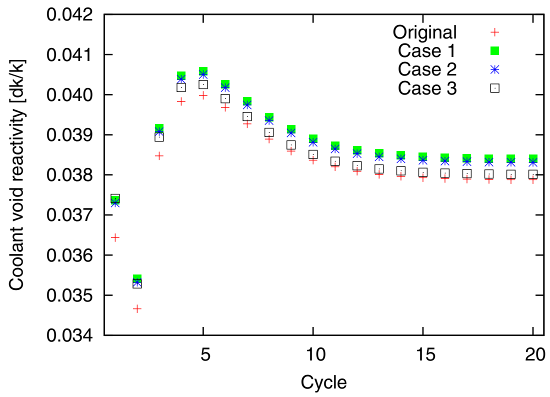

Figure 3. Coolant void reactivities.

their relative standard deviations before and after the adjustment. Chi-squares divided by the degree of freedom are 0.692 and 0.369 before and after the adjustment. Consistency among ND, the experimental data and the calculation values is confirmed from these results.

Next the neutronics parameters of the ADS at the beginning of cycles are calculated by the original and adjusted ND. Three sets of the adjusted ND are prepared: the first one, referred to as case 1, is prepared with the 31 integral data in which the lead-related data and the FCA-IX benchmark data are not included, the second one, case 2, is prepared with the 38 integral data, which are the case 1 data plus the lead-related data, and the third one, case 3, is prepared with all the integral data.

Figure 2 shows k_{eff}. It is interesting to point out that the adjusted ND set of case 3 gives larger values at the cycle 2 and the following cycles than other cases. This is because fission reaction rate ratios of Pu-238 to Pu-239 are underestimated with the original ND in the FCA-IX benchmark problem and this discrepancy is slightly improved by the ND adjustment.

On β_{eff}, there are almost no differences among the original and adjusted ND results.

Figure 3 shows the coolant void reactivity. Addition of the integral data related to the lead ND in the ND adjustment does not give any changes.

Finally ND-induced uncertainties of k_{eff}, β_{eff} and the coolant void reactivity at the beginning of cycles are shown in Fig. 4. The uncertainties of k_{eff} are reduced from $1.6\%\Delta k/k$ to $1.4\%\Delta k/k$ in a quasi-equilibrium state, and this reduction is attained with the 31 integral data. This uncertainty reduction might come from the adjustment of the fission cross sections of Np-237, Pu-238 and Am-241. However, this uncertainty reduction is not so large because the other ND such as capture cross sections, χ and $\bar{\nu}$, to which the integral data used in the present study are not sensitive, also contribute to total uncertainties. The uncertainties of β_{eff} and coolant void reactivity are not affected by the present ND adjustment; the integral data used in the present study do not contribute to the accuracy improvement of the ADS neutronics parameters. The reason why there is no impact of the ND adjustment on the coolant void reactivity is that inelastic scattering cross section of lead isotopes, to which the coolant void reactivity is sensitive, cannot be improved by the integral data presently used; the integral data related to lead isotopes (ID 32 to 38) are those of small and leaky fast neutron systems, so they have large sensitivities rather to cross section and angular distribution of elastic scattering which are significantly related to the neutron leakage.

6. Concluding remarks

We have quantified nuclear data-induced uncertainty of neutronics parameters of one specific ADS design. The nuclear data adjustment method with the integral data has been employed to reduce the uncertainties, and usefulness of the avaliable integral data has been investigated. Numerical results have revealed that the uncertainty reduction by the nuclear data adjustment with the available integral data is insignificant and restrictive.

Future perspecitives are as follows;

- Other integral data which are sensitive to reaction cross sections other than fission cross sections

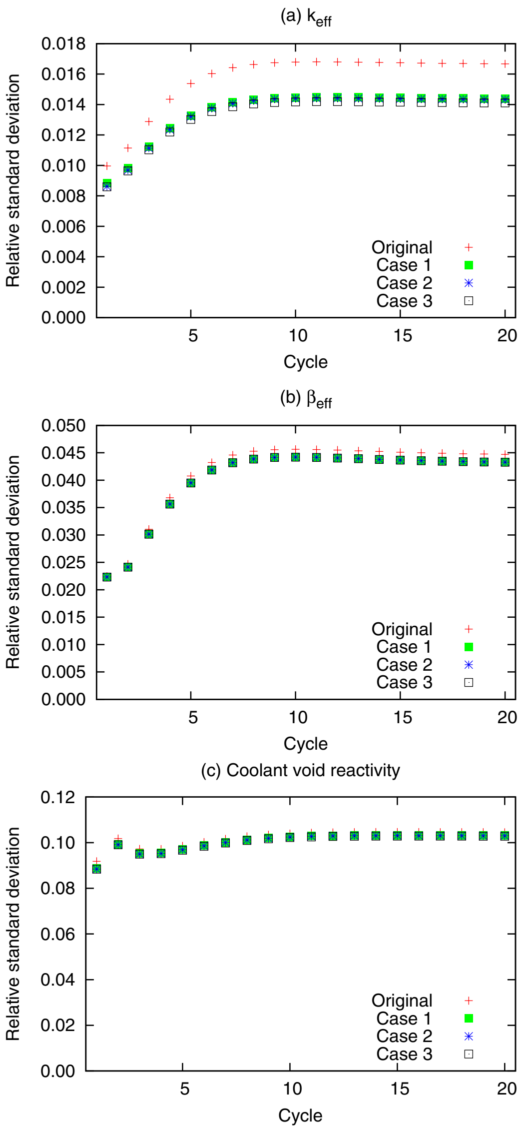

Figure 4. Nuclear data-induced uncertainties of k_{eff}, β_{eff} and coolant void reactivity.

should be utilized. Post irradiation examination data are promising candidates.

– On the uncertainty of β_{eff}, ν_d of Np-237 and Pu-238 are dominant contributors [8]. Although neutronics parameters in nuclear fission systems composed of Np-237 or Pu-238 as main fissile materials are expected to be sensitive to these ND, such experimental (or mock-up) systems are difficult to be realized. To improve these ND, efforts from the microscopic approach are quite important.
– To reduce the uncertainty of the coolant void reactivity, integral data sensitive to inelastic scattering cross sections of lead isotopes are beneficial.

This work is supported under the project *Basic Research for Nuclear Transmutation Techniques by Accelerator-Driven System*, a Special Research Program funded by Chubu Electric Power Co., Inc.

References

[1] M. Fukushima, A. Oizumi, H. Iwamoto, Y. Kitamura, JAEA-Data/Code 2014-030, Japan Atomic Energy Agency (2014) [in Japanese]
[2] M. Fukushima, K. Tsujimoto, S. Okajima, EPJ Web of Conference **111**, 07002 (2016)
[3] M. Fukushima, private communication
[4] K. Okumura, Y. Nagaya, JAEA-Data/Code 2011-010, Japan Atomic Energy Agency (2011) [in Japanese]
[5] S.C. van der Marck, Nucl. Data Sheets **113**, 2935–3005 (2012)
[6] M.B. Chadwick, et al., Nucl. Data Sheets **112**, 2887–2996 (2011)
[7] K. Tsujimoto, et al., J. Nucl. Sci. Technol. **41**, 21–36 (2004)
[8] G. Chiba, C.H. Pyeon, W. van Rooijen, T. Endo, J. Nucl. Sci. Technol. **53**, 1653–1661 (2016)